ACCESSION #: 9212180003
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Palo Verde Unit 2 PAGE: 1 OF 09

DOCKET NUMBER: 05000529

TITLE: Reactor Trip Due to Loss of CEDM Motor Generator Output
EVENT DATE: 11/13/92 LER #: 92-006-00 REPORT DATE: 12/14/92

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
OTHER

LICENSEE CONTACT FOR THIS LER:
NAME: Thomas R. Bradish, Manager, TELEPHONE: (602) 393-5421
Nuclear Regulatory Affairs

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On November 13, 1992, at approximately 1354 MST, Palo Verde Unit 2 was in Mode 1 (POWER OPERATION), operating at approximately 100 percent power, when the inadvertent deenergization of the Control Element Drive Mechanism (CEDM) Motor Generator (MG) Set "B" caused a reactor trip. The Control Element Assemblies (CEAs) inserted into the reactor core, causing a Main Turbine trip and the quick opening and modulation of the Steam Bypass Control System (SBCS) valves. This resulted in a change in steam flow causing primary system pressure to decrease below the low pressurizer pressure Engineered Safety Feature System actuation setpoint of 1837 psia. A valid actuation of the Safety Injection Actuation System (SIAS) and the Containment Isolation Actuation System (CIAS) occurred due to low pressurizer pressure.

This event was investigated in accordance with the APS Incident Investigation Program. The root cause of the reactor trip was determined to be the inappropriate actuation of the CEDM MG Set "B" remote motor

stop button by an auxiliary operator (AO). The root cause of the SIAS and CIAS was the over-cooling of the reactor coolant system caused by a loss of post-trip to SIAS operating margin for the SBCS. Pursuant to Technical Specification 3.5.2, ACTION b, this LER also provides the Special Report required for a Emergency Core Cooling System actuation.

There have been no previous similar events where the inappropriate actuation of a CEDM MG set caused a reactor trip reported pursuant to 10CFR50.73.

END OF ABSTRACT

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I. DESCRIPTION OF WHAT OCCURRED:

A. Initial Conditions:

At 1354 MST, on November 13, 1992, Palo Verde Unit 2 was in Mode 1 (POWER OPERATION) at approximately 100 percent power.

B. Reportable Event Description (Including Dates and Approximate Times of Major Occurrences):

Event Classification: An event that resulted in an automatic actuation of an Engineered Safety Feature (ESF) (JE) and the Reactor Protection System (RPS) (JE).

At approximately 1354 MST on November 13, 1992, Palo Verde Unit 2 experienced a reactor trip due to the inadvertent deenergization of the Control Element Drive Mechanism (CEDM) (AA) Motor Generator (MG) (AA) Set "B". The Control Element Assemblies (CEAS) (ROD) (AA) inserted into the reactor core, causing a Main Turbine trip. The Steam Bypass Control System (SBCS) (JI) responded as designed to the resulting change in steam flow and turbine pressure. The Core Protection Calculators (CPCS) (JE) generated Low Departure from Nucleate Boiling Ratio (DNBR) reactor trip signals due to CEA deviations which resulted from the CEA insertions, resulting in the opening of the reactor trip switchgear breakers (BKR) (AA). Primary system pressure decreased below the low pressurizer pressure Engineered Safety Feature Actuation System (ESFAS) (JE) setpoint of 1837 psia due to a loss of post-trip to Safety Injection Actuation System (SIAS) (JE) operating margin and subsequent overcooling by the SBCS. A valid actuation of the

SIAS and the Containment Isolation Actuation System (CIAS) (JE) occurred due to low pressurizer pressure. All safety systems functioned as required and the plant was stabilized in Mode 3 (HOT STANDBY) at approximately 1439 MST.

The CEDMs raise and lower the CEAs into the reactor core as a means of controlling core reactivity. The CEDM MG sets receive 480 VAC 3-phase power from non-vital power centers and transform it into reliable and constant motive power. Two identical 200 KW CEDM MG sets are provided, either of which can supply 100% of the power necessary to operate all of the CEDMs if necessary. Normally the CEDM MG sets operate in parallel sharing the load. Each CEDM MG set consists of a motor generator, two inertial flywheels, and related control equipment. Local control equipment is housed in the control cabinet (CAB) (AA). Each CEDM MG set can be remotely controlled from the local control panel (PL) (AA) of

the other CEDM MG set. Remote control is provided to allow the operator the ability to synchronize the MG sets during operation. The 3-phase motive power is supplied from the CEDM MG sets through the reactor trip switchgear to the CEDMS. The reactor trip switchgear provides a means of interrupting the source of power from the CEDM MG sets to the CEDMS. When the electromagnetic coils of the CEDMs are de-energized, the CEAs fall into the core by gravity to shutdown the reactor.

Prior to the reactor trip on November 13, 1992, CEDM MG Set 'A' was taken out of service by an auxiliary operator (AO) (utility, non-licensed) to perform scheduled 6-month interval preventive maintenance (PM) in accordance with approved procedures. CEDM MG Set "B" was supplying power to all the CEDMS. At approximately 1240 MST, the AO was directed by the control room to restart CEDM MG Set "A" and to run the set unloaded for one hour. When the hour run was complete the AO returned to the CEDM MG set control panel to shut down the CEDM MG Set "A". The AO located CEDM MG Set "A" motor stop button and turned to look at his watch to note the exact time. When he returned to the CEDM MG set control panel he incorrectly placed his finger on the CEDM MG set "B" remote motor stop button resulting in the loss of power to the CEDMs and subsequent CEA insertion into the reactor core. The CEDM MG Set "B" Input Breaker Open alarm annunciated in the Control Room. Approximately six seconds later the CEDM MG Set "B"

Output Breaker Open alarm annunciated in the Control Room.

At approximately 1354 MST the Main Turbine tripped due to a low voltage signal received as a result of the deenergization of the CEDM circuitry. The CPCs generated low DNBR reactor trip signals due to CEA deviations which resulted from the CEA insertions, resulting in the opening of the reactor trip switchgear breakers. Normally, this would cause the CEAs to insert into the reactor core, shutting down the reactor. However, due to the loss of power to CEDM MG set "B", the CEAs were already inserting. The SBCS subsequently responded as designed to the resulting change in steam flow and turbine pressure with a quick open signal to four of the SBCS valves (PCV) (JI). The SBCS valves then modulated as required to control the secondary side pressure.

The SBCS is designed to automatically control the plants' secondary system response to a reduction in turbine capacity. The SBCS provides a means for controlling primary system thermal conditions during heatup, cooldown and after unit trips by controlling steam flow demand on the steam generators. Valve modulation is used to control relatively slow primary system transients. On a reactor or turbine trip from high power, the SBCS maximizes energy release by initially quick opening four or

eight valves, as required, based on the size of the load rejection. After excess energy is dissipated, the SBCS modulates all of the valves, as required, to maintain the main steam header pressure at its hot standby conditions.

The opening of the SBCS valves caused primary system temperature to decrease resulting in a pressure decrease below the low pressurizer pressure ESFAS setpoint of 1837 psia. A valid actuation of the SIAS and the CIAS occurred approximately 37 seconds after the reactor and turbine trip due to low pressurizer pressure. A small amount of injection flow is assumed to have occurred due to the reactor coolant system pressure dropping below the shutoff head of the High Pressure Safety Injection pumps (P) (BQ).

By approximately 1409 MST, Control Room personnel stopped Trains A and B Containment Spray pumps (P) (BQ), High Pressure Safety Injection pumps, and Low Pressure Safety Injection pumps

(P) (BE). The plant was stabilized in Mode 3 (HOT STANDBY) at approximately 1439 MST. At approximately 1517 MST, following verification of proper safety system actuation, the SIAS and CIAS were reset.

The Emergency Plan Implementing Procedures require the declaration of a Notification of Unusual Event (NUE) for an event resulting in a SIAS actuation, caused by a valid low pressurizer pressure condition. An NUE was declared at approximately 1405 MST and subsequently terminated at approximately 1439 MST.

C. Status of structures, systems, or components that were inoperable at the start of the event that contributed to the event:

Other than CEDM MG Set "A" undergoing preventive maintenance as described in Section I.B. no structures, systems, or components were inoperable at the start of the event which contributed to this event.

D. Cause of each component or system failure, if known:

Not applicable - no component or system failures were involved.

E. Failure mode, mechanism, and effect of each failed component, if known:

Not applicable - no component failures were involved.

F. For failures of components with multiple functions, list of systems or secondary functions that were also affected:

Not applicable - no failures of components with multiple functions were involved.

G. For a failure that rendered a train of a safety system inoperable, estimated time elapsed from the discovery of the failure until the train was returned to service:

Not applicable - no failures that rendered a train of a safety system inoperable were involved.

H. Method of discovery of each component or system failure or procedural error:

Not applicable - there have been no component or system failures identified.

I. Cause of Event:

An independent investigation of this event was conducted in accordance with the APS Incident Investigation Program. The investigation determined that the auxiliary operator responsible for inadvertent actuation of the CEDM MG Set "B' remote motor stop button had the appropriate procedures available and in use and had properly completed the requirements necessary to perform the preventive maintenance on the CEDM MG Set "A". However, due to momentary lapse of self-verification techniques, specifically to verify that the correct labeled component number was selected prior to operation, the CEDM MG Set "B" remote motor stop button was pressed. Hence, the root cause of the reactor trip was determined to be the inappropriate actuation of the CEDM MG Set 'B' remote motor stop push button by an auxiliary operator (AO), which deenergized the CEDMs and caused the insertion of the CEAs (SALP Cause Code A: Personnel Error).

The SBCS responded as designed to the reduction in steam flow and turbine pressure with a quick open signal to four of the SBCS valves, and with the SBCS valves modulating as required to control secondary pressure. A previous study which led to recent modifications to optimize the SBCS had predicted a minimum reactor coolant system pressure of above 1925 psia following an uncomplicated trip from full power operating conditions. This results in a post-trip margin for SIAS actuation of approximately 88 psia. The minimum post-trip pressure for this event was approximately 1822 psia (approximately 100 psia from the predicted value). Hence, the root cause of the SIAS and CIAS was

overcooling of reactor coolant system caused by a loss of post-trip to SIAS margin for the SBCS. Computer simulations were performed and actual plant data for the event was analyzed to identify the factors which contributed to the overcooling. The overcooling was determined to be due to a combination of the following factors:

Pressurizer pressure at the start of the event was approximately 33 psia below the nominal 2250 psia value assumed during a study which was previously performed to optimize the SBCS response. The plant was operating at a slightly reduced pressure of 2217 psia due to a pressurizer safety valve (PSV) (PZR) (RV) (AB) elevated tailpipe temperature. Elevated tailpipe temperatures can be caused by a number of factors. Weepage past the isolation valve between the Reactor Drain Tank (RDT) (TK) (CA) and the Reactor/Pressurizer Head Vent System (AB) can raise the temperature in the common tailpipe header. Leakage past the Regenerative Heat Exchanger drain valves (V) (CB) may also raise the RDT temperature resulting in an increased tailpipe temperature. Thus, differentiation between a leaking PSV, leaking isolation valve, or other sources is difficult. Based on plant and industry operating experience, it has been determined that reduced RCS pressure assists in controlling this problem. Consequently, Unit 2 was operating at the reduced pressure. Operation at a reduced pressure directly correlates to a reduction in the margin to post-trip SIAS.

The average time for the SBCS valves to modulate closed following the trip was slightly longer than nominal closure time. The valves modulating closed more slowly resulted in additional energy removal from the steam generators which accounts for an additional loss in the margin to post-trip SIAS actuation. However, the valves did operate as designed.

The sequence of the trip scenario (rods start to fall and turbine trip at essentially the same time) also contributed to the overcooling since the RCS began cooling rapidly prior to the decrease in secondary side steam demand.

An additional loss in margin can be attributed to the scram worth curves previously used in the simulations from the optimization study. These curves

conservatively assume an axial shape index which results in very little negative reactivity insertion until the CEAs are nearly fully inserted. The total scram worth is also conservatively small. When best estimate values are used, the simulated reactor power drops more quickly after a reactor trip, with a minimum pressurizer pressure of approximately 20 psia lower than previously predicted.

The sum of all these factors accounted for the post-trip SIAS actuation for the event. Based on the results of the actual plant data and comparison to the simulation results, the SBCS functioned as expected. The loss of post-trip to SIAS operating margin was the major contributor to the SIAS actuation.

J. Safety System Response:

The following safety systems actuated as a result of the event:

- High Pressure Safety Injection (BQ), Trains A and B
- Low Pressure Safety Injection System (BQ), Trains A and B
- Containment Spray System (BE), Trains A and B,
- Emergency Diesel Generators (EK), Trains A and B,
- Essential Chilled Water System (KM), Trains A and B
- Essential Cooling Water System (BI), Trains A and B,
- Essential Spray Pond System (BS), Trains A and B
- Condensate Transfer System (KA), Trains A and B,
- Control Room Essential Heating, Ventilation and Air Conditioning (HVAC) System (AHU)(VI), Trains A and B,
- Auxiliary Building Essential HVAC System (AHU)(VF), Trains A and B,
- Fuel Building Essential HVAC System (AHU)(VG), Trains A and B,
- Engineered Safety Features Switchgear Essential HVAC System (AHU)(VJ), Trains A and B,
- Containment Isolation System (JM), and
- Auxiliary Feedwater Pump (P) (BA), Train B

The reactor protection system operated within the Technical Specification Limit of
DNBR trip signals on the Plant Protection System (PPS)

K. Failed Component Information:

Not applicable - no component failures were involved.

II. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS

EVENT:

A safety limit evaluation was performed as part of the APS Incident Investigation. The evaluation determined that the plant responded as designed, that no safety limits were exceeded, and that the event was bounded by current safety analyses.

Nuclear Fuel Management (NFM) performed an assessment of the event and determined that the equipment and systems assumed in the Updated Final Safety Analysis Report (UFSAR) Chapter 15 were functional and performed as required. No abnormal transients were identified following the reactor trip. Scenarios defined in UFSAR Chapter 6 concerning loss of coolant accidents (LOCAS) were not challenged during this event.

The assessment concluded that the event did not result in a transient more severe than those already analyzed. All safety systems functioned as required. The event did not result in any challenges to the fission product barriers or result in any releases of radioactive materials. Therefore, there were no adverse safety consequences or implications as a result of this event. This event did not adversely affect the safe operation of the plant or the health and safety of the public.

III. CORRECTIVE ACTION:

A. Immediate:

An investigation team was formed and an investigation was initiated in accordance with the APS Incident Investigation Program. As part of the investigation, APS initiated a root cause investigation.

B. Action to Prevent Recurrence:

The auxiliary operator was disciplined in accordance with the PVNGS Positive Discipline Program.

Identification and quantification of the contributing factors to the post-trip overcooling was performed using computer simulations. Actual plant data was reviewed and evaluated and the contributing factors described in Section I.1 were identified.

The root cause of failure for the SIAS and CIAS was the overcooling of the reactor coolant system by the SBCS, caused by the loss of post-trip to SIAS operating margin. APS is evaluating potential actions that will further optimize the SBCS and increase the post-trip to SIAS operating margin.

IV. PREVIOUS SIMILAR EVENTS:

There have been no previous similar events reported pursuant to 10CFR50.73.

V. ADDITIONAL INFORMATION:

Based on reviews by the Plant Review Board, the Management Response Team, and the Unit 2 Plant Manager, a Unit restart was authorized in accordance with approved procedures. Unit 2 entered Mode 2 (STARTUP) at approximately 0336 MST on November 15, 1992 and Mode 1 (POWER OPERATION) at approximately 1130 MST on November 15, 1992. Unit 2 synchronized to the grid at approximately 1507 MST on November 15, 1992.

The event was simulated using the SBCS setpoints and configuration prior to the modifications aimed at optimizing SBCS performance. The major differences are that the unmodified SBCS did not include modulation signal tracking of the quick open demand and the unmodified SBCS would have quick opened seven (the number of SBCS valves in-service prior to the modification) instead of four valves. The purpose of this effort was to determine whether the modified SBCS acted to increase or decrease the severity of this event as compared to the unmodified SBCS configuration. The simulation results indicate that the unmodified SBCS would have resulted in significantly lower minimum RCS pressures than actually occurred with the optimized SBCS.

VI. SPECIAL REPORT:

In Palo Verde Unit 2, there have been 6 total accumulated actuation cycles of the Emergency Core Cooling System to date. This satisfies the requirements of Technical Specification 3.5.2 ACTION b.

ATTACHMENT 1 TO 9212180003 PAGE 1 OF 1

Arizona Public Service Company
PALO VERDE NUCLEAR GENERATING STATION
P.O. BOX 52034 o PHOENIX, ARIZONA 85072-2034

JAMES M. LEVINE
VICE PRESIDENT 192-00815-JML/TRB/NLT
NUCLEAR PRODUCTION December 14, 1992

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Mail Station P1-37
Washington, D.C. 20555

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 2
Docket No. STN 50-529 (License No. NPF-51)
Licensee Event Report 92-006-00
File: 92-020-404

Attached please find Licensee Event Report (LER) 92-006-00 prepared and submitted pursuant to 10CFR50.73. This LER reports a Unit 2 reactor trip and a valid actuation of the Safety Injection Actuation System and the Containment Isolation Actuation System. In accordance with 10CFR50.73(d), a copy of this LER is being forwarded to the Regional Administrator, NRC Region V.

Pursuant to Technical Specification 3.5.2, ACTION b, this LER also describes the actuation of the Emergency Core Cooling System, and satisfies the Special Report requirements for documenting the total accumulated actuation cycles to date for Unit 2.

If you have any questions, please contact T. R. Bradish, Manager, Nuclear Regulatory Affairs at (602) 393-5421.

Very truly yours,

JML/TRB/nlt

Attachment

cc: W. F. Conway (all with attachment)
J. B. Martin
J. A. Sloan
INPO Records Center

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